**Proceedings of ICONE 10:
International Conference on Nuclear Engineering
April 14-18, 2002 Arlington, Virginia**

**ICONE10-22641**

# TRANSIENT ANALYSIS NEEDS FOR GENERATION IV REACTOR CONCEPTS

L. J. Siefken, Idaho National Engineering and Environmental Laboratory

E. A. Harvego, Idaho National Engineering and Environmental Laboratory

E. W. Coryell, Idaho National Engineering and Environmental Laboratory

C. B. Davis, Idaho National Engineering and Environmental Laboratory

## ABSTRACT

The importance of nuclear energy as a vital and strategic resource in the U. S. and world's energy supply mix has led to an initiative, termed Generation IV by the U.S. Department of Energy (DOE), to develop and demonstrate new and improved reactor technologies. These new Generation IV reactor concepts are expected to be substantially improved over the current generation of reactors with respect to economics, safety, proliferation resistance and waste characteristics. Although a number of light water reactor concepts have been proposed as Generation IV candidates, the majority of proposed designs have fundamentally different characteristics than the current generation of commercial LWRs operating in the U.S. and other countries. This paper presents the results of a review of these new reactor technologies and defines the transient analyses required to support the evaluation and future development of the Generation IV concepts. The ultimate objective of this work is to identify and develop new capabilities needed by INEEL to support DOE's Generation IV initiative. In particular, the focus of this study is on needed extensions or enhancements to SCDAP/RELAP5/3D code. This code and the RELAP5-3D code from which it evolved are the primary analysis tools used by the INEEL and others for the analysis of design-basis and beyond-design-basis accidents in current generation light water reactors.

## 1. INTRODUCTION

At this stage in the development of Generation IV reactors, licensing analysis requirements for these reactors have not been completely defined by the United States Nuclear Regulatory Commission (USNRC). However, just as calculations are required of the transient behavior of the current generation of reactors following a broad range of initiating events (USNRC 1981), so similar requirements are expected for Generation IV reactors. For the current generation of reactors, the initiating events required for consideration include; (1) break in coolant system piping, (2) anticipated transient without scram, (3) ejection of reactor control rods, (4) inadvertent opening of a valve, (5) break in the shaft of a coolant pump, (6) startup of an inactive coolant loop, and (7) loss of off-site power (Greene et al 2001). In general, these initiating events are any event that adversely perturbs the reactor during its normal state of power production. While these general categories of initiating events and the resulting transient responses may not be appropriate for all of the different Generation IV reactor concepts, they do provide a basis for assessing the vulnerability of the different designs to recognized potential accident initiating events, and provide a starting point for the evaluation of other potential initiating events that may be unique to a particular design concept.

The following sections describe current INEEL analysis capabilities, discuss the four general categories of Generation IV reactor concepts considered in this study, their analysis needs, and the transient analysis capabilities required to address these needs. Specifically, Section 2 of this paper describes the current capabilities of SCDAP/RELAP5/3D as they are applied to current generation reactors. Section 3 provides a brief description of representative designs in four general categories of Generation IV reactors. Section 4 presents a more detailed description of the designs of reactors in each category, their requirements for transient analyses, and the extensions in modeling capability required. Section 5 then summarizes the extensions required to SCDAP/RELAP5/3D for the transient analyses of Generation IV reactors, and Section 6 presents the conclusions of this study.

## 2. DESCRIPTION OF SCDAP/RELAP5/3D CODE

The SCDAP/RELAP5/3D code is an extension of the RELAP5-3D computer code (RELAP5-3D, 2001), which has been developed primarily for the thermal-hydraulic analysis of light water nuclear reactors and related experimental systems. The RELAP5-3D code can simulate a wide variety of thermal-hydraulic transients involving steam, water, and non-condensable fluid mixtures. These transients include design-basis loss-of-coolant accidents and operational events in commercial pressurized and boiling water reactors, test and production reactors operated by the Department of Energy, reactors designed by the former Soviet Union, and related experimental systems.

The components of a nuclear reactor are represented with a user-defined nodalization that contains hydrodynamic control volumes and junctions that represent flow paths between control volumes, and heat structures. The code solves separate continuity, momentum, and energy equations for the gas and liquid phases. Each phase can have a different temperature and velocity within a control volume. The code contains heat conduction and wall heat transfer models to simulate the energy exchange between structures and hydrodynamic control volumes. The code contains models to represent the nuclear kinetics of a reactor core. The code also contains a flexible control system that allows the user to represent physical control systems within plants.

The RELAP5 code (RELAP5, 1995) was originally developed for one-dimensional applications and utilized one-dimensional hydrodynamic and heat conduction models and a point reactor kinetics model, in which the spatial distribution of power generated by the reactor remained fixed. The RELAP5-3D code has been improved and contains multi-dimensional hydrodynamic and reactor kinetics models.

The SCDAP/RELAP5/3D code (Coryell et al, 2001) has been developed to allow calculation of light water nuclear reactor system response for beyond-design-basis accidents, including core damage. The code is the result of merging RELAP5-3D with the SCDAP models in the SCDAP/RELAP5/MOD3.3 code (Siefken et al 2001). The reactor systems can be modeled using an arbitrary number of fluid control volumes and connecting junctions, core components and system components. Flow areas, volumes, and flow resistances can vary with time through either user-control or models that describe the changes in geometry associated with damage in the core. System structures can be modeled with RELAP5 heat structures, SCDAP two-dimensional core components, or SCDAP two-dimensional porous medium models. The SCDAP core components include representative light water reactor fuel rods, silver-indium-cadmium (Ag-In-Cd) and $B_4C$ control rods and/or blades, electrically heated fuel rod simulators, and structures such as reactor vessels and concrete structures. Other system components available to the user include pumps, valves, electric heaters, jet pumps, turbines, separators and accumulators.

## 3. RANGE OF DESIGNS PROPOSED FOR GENERATION IV

Generation IV reactor designs can be grouped into four general categories. These four general categories are (1) gas-cooled reactors (Southworth et al 2001), (2) liquid-metal-cooled reactors (Rosen et al 2001), (3) non-classical reactors such as those with molten salt coolant or gas cores (Anghaie and Lewis 2001), and (4) light water reactors (MacDonald et al 2001). After reviewing a number of reactor designs in each of these categories, it was concluded that representative designs in each of these general categories demonstrated characteristics that were common to most of the designs in a particular category. This observation led to the decision to focus the evaluation of transient analysis needs on one or two representative designs in each of the four general reactor categories. This approach allowed for an in-depth evaluation of representative designs in each category, while at the same time making the task of defining general analysis requirements more manageable.

In the category of gas-cooled reactor concepts, a pebble bed reactor (Brey 2000, Brey 2001, Yamashita 1990, Gittus 1999, McNeill 2001), and a graphite-block reactor (McCardell et al 1990, DOE 1994, Kunitomi et al 1998, IAEA 1997) were selected for more detailed evaluation. These reactors are candidates for Generation IV reactors because they offer a very efficient conversion of nuclear power to electrical power as well as inherent safety features. Both reactor designs use direct Brayton cycle gas turbines for electric power generation and have an energy conversion efficiency of 40% to 45%. The Brayton cycle was selected for both concepts over the traditional Rankine cycle because the higher potential cycle efficiency is thought to more likely meet the “Competitive Busbar Cost” goal for Generation IV reactors. The graphite moderator material and gas (helium) coolant for the two selected reference designs are a common feature for all the designs in this category, and are the most important features in determining analysis requirements for reactors in this category.

In the category of liquid-metal-cooled reactors, both lead-bismuth (Pb-Bi) (MacDonald et al 2000, Sekimoto and Su’ud 1994, Spencer et al 2000, Weaver et al 2001), and sodium-cooled (Boardman 2000) reactors were chosen for more detailed evaluation. As will be discussed later, the need to evaluate the two liquid-metal coolants was necessary because of the vastly different properties and behavior of these coolants under normal and transient operating conditions. Since these two basic designs may have several different fuel composition and coolant configurations, evaluations were also made of the affect of these design parameters on reactor transient response and operating limits.

In the category of non-classical reactors, both molten salt reactors and reactors with gas cores have been proposed. One proposed molten salt reactor has the fuel dissolved in a salt so as to have a liquid core with graphite moderator elements (Robertson 1971, Gat and Dodds 1997, Mitachi et al 1999). Another proposed molten salt reactor has molten salt as the coolant and the fuel configured as coated fuel particles in graphite blocks (Vergnes et al 2000). The gas core concept utilizes magneto-hydrodynamics for electrical power production and claims an energy conversion efficiency greater than 60%. Although a number of non-classical reactors have been proposed as potential Generation IV concepts the viability of these reactor designs remains to be shown. However, these designs are being proposed because they offer the possibility of a very high thermal efficiency as well as having proliferation resistant attributes. For these reasons, and because of some very unique analysis requirements, two non-classical reactor concepts were selected for more detailed evaluation.

A number of light water reactor designs have been proposed as Generation IV candidates. Among these designs are (1) Supercritical Pressure Fast Reactor (SPWR) (Oka et al 1995, Jevremovic et al 1996, Kitoh et al 1998, Mukohara et al 1999, MacDonald et al 2001), (2) International Reactor Innovative and Secure (IRIS) (Carelli et al 2000, Carellli et al 2001), (3) Simplified Boiling Water Reactor (SBWR) (Upton et al 1993, Ishii 1999, Rao and Gonzalez 2000, Brettschuh 2001), (4) Multi-Application Small Light Water Reactor (MASLWR) (Modro et al 2000), and (5) CANDU Supercritical Pressure Water Reactor (Busby et al 2000). Although many of the transient analysis capabilities developed for the analysis of current generation light water reactors (LWRs) can be utilized in the analysis of these various Generation IV LWR concepts, LWRs operated above the critical pressure of water present some unique challenges to our current analysis capabilities, as will be discussed later in this paper.

The basic features or characteristics of several representative Generation IV reactor designs in each of the four general categories are described in Table 1. The abbreviations used in this table are defined in Table 2.

## 4. TRANSIENT ANALYSIS REQUIREMENTS FOR VARIOUS CATEGORIES OF GENERATION IV REACTORS

The licensing of any proposed new nuclear reactor concept will require the evaluation of the behavior of the reactor over a broad range of potentially adverse events. Although the specific analysis requirements may vary, as a minimum, new designs will be required to demonstrate acceptable response characteristics for a broad range of potential initiating events similar to those considered in the licensing of current generation light water reactors.

The following sections of this paper describe some of the features of reactors within the four general categories of Generation IV reactors, potential transients events relating to the safety of these concepts, and the analysis capabilities needed to address these events.

### 4.1 Gas-Cooled Reactors

Two basic HTGR core designs were considered in this study. The first type, namely the PB-HTGR, has the fuel configured as a deep porous bed of pebbles through which the helium coolant is forced (Yamashita 1990, Gittus 1999, Brey 2001, McNeill 2001). The second type of core design, namely the block-type or prismatic-type, has the fuel configured as rods inside blocks of graphite (McCardell et al 1990, DOE 1994, Kunitomi et al 1998, IAEA 1997). A representative design for the pebble-bed reactor is shown in Figure 1, and the component configuration of a graphite-block core design is shown in Figure 2.

For the pebble bed HTGR, the fuel pebbles are spherical in shape and have a diameter of about 60 mm. Each fuel pebble is composed of many small particles of uranium dioxide coated with carbon and silicon carbide. The coating on the fuel particles retains fission products. The fuel particles are placed in a matrix of graphite. Fuel pebbles can be continuously fed into the top of the operating reactor and then move downward through the reactor core region. The block-type HTGR has similar coated fuel particles in a matrix configured as rods instead of as pebbles. The rods of fuel particles are placed in graphite blocks.

The HTGRs in this study utilized the direct Brayton thermodynamic cycle. In this thermodynamic cycle, the helium heated by the reactor fuel flows out of the bottom of the reactor vessel, then drives a set of turbo-compressors before expanding into the main turbine, which is shaft connected to the generator. The expanded helium then passes through a regenerative heat exchanger and a water-cooled precooler before being repressurized in a set of turbo-compressors, regeneratively heated, and returned to the top of the reactor vessel. This thermodynamic cycle yields energy conversion efficiency in the range of 40% to 45%. The pressure of the helium in the reactor vessel is 7 MPa. The temperature of the helium entering the top of the core is 775 K and its temperature exiting the bottom of the reactor core is 1170 K.

Passive cooling systems have been designed for both the pebble bed and block-type HTGR. For the pebble-bed HTGR, decay heat is transferred by conduction, radiation and natural convection to the ground around the reactor building. The reactor building is designed to limit the ingress of air and the oxidation of graphite in the event of the rupture of a pipe (Gittus 1999). For the block-type HTGR, decay heat is transferred by conduction to the outer surfaces of the reactor core and then by radiation and natural convection to the reactor vessel (Kunitomi 1998). In one design of the block-type HTGR, the heat is removed from the reactor vessel by conduction, radiation, and natural circulation to the atmosphere. In another design, the reactor vessel is cooled by water surrounding it.

Table 1. Features of various proposed Generation IV reactors.

| Feature of Reactor | Proposed Generation IV Reactor | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | Light water cooled | | | Gas cooled | | Liquid metal cooled | | Non-classical |
| | SBWR, MASLWR | IRIS | SPWR, CANSP | PB-HTGR | BT-HTGR | HMFR | S-PRISM | MSR, GCR |
| Composition of primary coolant | $H_2O$ | $H_2O$ | $H_2O$ | He | He or $CO_2$ | Pb-Bi | Na | Sa, $UF_4$ |
| Coolant pressure (MPa) | 6.9, 8 | 17.2 | 25.0 | 7.0 | 7.0 | 0.1 | 0.1 | 0.5, 6.0 |
| Coolant temperature at core exit (K) | 558, 567 | 603 | 780, 898 | 1170 | 1125 | ~850 | 783 | 980-1273, 2500 |
| Composition of fuel | $UO_2$ | $UO_2$ or MOX | MOX, $UO_2$-$ThO_2$ | $UO_2$ | $UO_2$ | Pu-Zr, UN | MOX | Th-U, U |
| Irradiation cycle period (years) | >2, >3 | 5-10 | ~2, >2 | 3 | >5 | >5 | 2 | ?, ? |
| Configuration of fuel and surrounding material | Rod/f | Rod/f | Rod/f | Ball/C | Rod/C | Rod/f | Rod/f | Liquid or Rod/C, Vapor |
| Neutron spectrum | mod | mod or hard | hard, mod | mod | mod | hard | hard | mod and hard, ? |
| Location of primary coolant loop | In-ves | In-ves | Ex-ves, Ex-ves | Ex-ves | Ex-ves | In-ves | In-ves | Ex-ves, Ex-ves |
| Driving force for primary coolant | NC | pump & NC | pump, pump | tur-cmp | tur-cmp | pump or NC | pump | pump, pump |
| Thermal cycle | In-Dir except SBWR | In-Dir | Dir, Dir | Dir | Dir | In-Dir | In-Dir | In-Dir, Dir |
| Thermal efficiency (%) | 34, 23 | ~30 | 45, 41 | 40 | 40 | 39 | 40 | 44, >60 |
| Passive transfer of shutdown decay heat to large pool of water, air, or earth? | yes, yes | yes | no, no | yes | yes | yes | yes | no, ? |
| Active emergency cooling system? | yes, yes | yes | yes, yes | no | no | no | no | no, no |
| Minimum size of reactor (MWe) | 600, 35 | 100 | 1500, 500 | 100 | 100 | 100 | 380 | 1000, 70 |
| Containment Building? | Inert, Yes | Inert, pres | Yes, Yes | No | ? | Yes | Yes | Yes, Yes |

**Table 2. Definition of abbreviations used in Table 1.**

| Abbreviation | Definition |
|---|---|
| Ball/f | 400,000 fuel particles with diameter of 0.5 mm, each coated with graphite, contained in 60 mm ball with graphite matrix, and outer surface of ball in contact with coolant |
| BT-HTGR | Block Type – High Temperature Gas Reactor |
| CANSP | CANDU Supercritical Pressure Water Reactor |
| Dir | Direct power cycle, electricity generating turbine driven by primary fluid, no steam generator |
| ex-ves | Part of primary coolant loop, such as steam generators or pumps, are outside of reactor vessel |
| GCR | Gas Core Reactor |
| HMFR | Heavy Metal Fast Reactor |
| In-Dir | Indirect power cycle, electricity generating turbine driven by fluid other than primary fluid, steam generator used |
| In-ves | Entire primary coolant loop is inside reactor vessel |
| IRIS | International Reactor Innovative and Secure |
| liq | Liquid |
| MASLWR | Multi-Application Small Light Water Reactor |
| mod | Moderated (thermal) neutron flux |
| MOX | Mixture of $UO_2$ and $PuO_2$ |
| MSR | Molten Salt Reactor |
| NC | 100% of driving force for primary coolant is supplied by buoyancy force (natural circulation) |
| p | Pressure (MPa) |
| Pres | Pressurized |
| PB-HTGR | Pebble Bed – High Temperature Gas Reactor |
| Rod/C | Rod-shaped compact of fuel particles with diameter of 0.47, each fuel particle coated with layer of Arc or other material, and compact embedded in graphite block, and flow channels for coolant inside block. |
| Rod/f | Fuel with cladding in configuration of rod and outer surface of rod in contact with fluid |
| Sa | Fuel salt composed of $LiF$-$BeF_2$-$ThF_4$-$UF_4$ or some variation of this basic combination |
| SBWR | Simplified Boiling Water Reactor |
| S-PRISM | Sodium-cooled Fast Reactor |
| SPWR | Supercritical Pressure Water Reactor |

The evaluation of the safety of a HTGR requires the capability to calculate the transient behavior of the reactor following a broad range of initiating events. While the designs of HTGRs are radically different than those of LWRs, they are nevertheless vulnerable to accident initiating events similar to those occurring in a LWR.

A pipe break in an HTGR results in the possibility for ingress of water or air into the reactor core and thus the possibility of oxidation of graphite components in the reactor core (DOE 1994). The ingress of liquid water into the reactor core enhances neutron moderation and increases the reactivity of the core and the possibility of an excursion in reactor power. The ingress of air or steam into the reactor core causes oxidation and heat up of graphite and the hydrolysis of initially failed fuel particles. The oxidation of graphite releases fission products trapped in the graphite and reduces the strength of the graphite. The hydrolysis of failed fuel particles results in fission product release from these particles. For the direct Brayton-cycle designs, the sources for moisture ingress include the precooler, intercooler, and the shutdown cooling system heat exchangers. Thus, the transient analyses of HTGRs involve the calculation of the distance of penetration of water/air into the core.

Another safety concern for the direct Brayton-cycle HGTR is a possible failure of the gas turbo machinery (DOE 1994). The history of gas turbine operations indicates that parts from a gas turbine may break off and block flow through the turbine passages. The blockage of flow through the turbine could result in a large axial pressure drop that damages the core support structure and challenges the safety functions of heat removal and control of core heat generation. A blockage could also cause a reversal of flow through the core, which in turn may cause an ejection of control rods and challenge control of core heat generation (DOE 1994). Transient analyses of the behavior of a direct cycle HTGR following a turbine deluding event have been performed using the RELAP5/MOD3 code (DOE 1994).

While the SCDAP/RELAP5/3D code has the capability to model basic phenomena occurring in a HTGR after an accident-initiating event (DOE 1994), nevertheless some features of the HTGR cannot be adequately represented by the code. For example, the calculation of the transient behavior of the reactor core of a HTGR requires several extensions in the current modeling to adequately account for (1) conduction of heat through spherical fuel/graphite pebbles, (2) convective heat transfer and flow losses in the bed of pebbles, (3) oxidation of pebbles, and (4) heat transfer by conduction, radiation, and natural convection through a bed of pebbles to the outer surfaces of the reactor core. For the block-type HTGR, these extensions in modeling are similar, except that extensions are required for modeling the removal of decay heat by conduction through the graphite blocks to the outer surfaces of the reactor core.

The calculations of the transient thermal hydraulic behavior of a HTGR also requires the capability to calculate multidimensional fluid behavior (Schultz 2001, van Heek 2001), and to calculate the ingress of air or water into the reactor vessel following a break in a pipe or other component. For accident initiating events with either forced flow or loss of forced flow, the possibility exists for local deficiencies in cooling that may result in hot spots in the reactor core. The identification of these local deficiencies in cooling and hot spots requires the application of a multidimensional Computational Fluid Dynamics (CFD) code. In an event resulting in complete loss of forced flow, a significant part of the removal of decay heat from the reactor core will occur by multidimensional natural circulation of the gas remaining in the reactor vessel. The modeling of this natural circulation of gas may also require the application of a CFD code. In the event of a pipe break, a jet of hot gas may impinge on the structure near the pipe break. The modeling of this jet of gas and the temperature and pressure loads it applies to the impinged structure may also require the application of a CFD code (van Heek 2001). These applications of a CFD code can be achieved by interfacing the SCDAP/RELAP5/3D code with CFD codes such as FLUENT (Freitas 1995, Schultz 2001).

Fundamental measures of the safety of an HTGR are the calculated temperature histories of the reactor core and vessel following any adverse event. The capability of the HTGR reactor core to endure high temperatures allows the design of a passive decay heat removal system involving the transfer of heat by conduction, radiation and natural convection from the inner parts of the reactor core to the inner surface of the reactor vessel, and then heat transfer by these same mechanisms from the reactor vessel to the environment beyond the reactor building. The calculation of this transfer in heat may require multidimensional modeling with temperatures calculated at up to 50,000 different locations in the reactor and its surrounding environment (Kadak 2001). The capability for modeling this heat transfer could be achieved by interfacing SCDAP/RELAP5/3D with a multidimensional heat transfer code such as HEATING7 (Kadak 2001), but may also be possible with the current SCDAP/RELAP5/3D multi-dimensional modeling capabilities.

## 4.2 Liquid-Metal Cooled Reactors

Liquid metal cooled fast reactors have been proposed as candidates for Generation IV reactors because they have a high thermal efficiency as well as having a fast neutron spectrum for burning actinides. The high boiling temperature of the liquid metal coolant permits high coolant temperatures, which in turn yields high overall plant efficiency (~40%) under very low system pressure. The high boiling temperature of the liquid metal coolant also has the advantage of counteracting the positive void reactivity coefficient of these reactors.

The majority of liquid metal reactors evaluated in this study used either sodium (Na) or a lead-bismuth eutectic (Pb-Bi) as the coolant. The design of the S-PRISM sodium cooled reactor (Boardman et al 2000) is shown in Figure 3 and the design of the STAR-LM Pb-Bi cooled reactor (Spencer et al 2000) is shown in Figure 4. The performance of sodium cooled reactors has been demonstrated by reactors such as the Integral Fast Reactor (Weaver et al 2001). Sodium and Pb-Bi coolants have advantages and disadvantages. Sodium coolant has the advantages of (1) not being corrosive to structural materials, (2) having a relatively low materials cost, and (3) being based on well-established technologies. Sodium coolant has the disadvantages of (1) reacting energetically with water, (2) having a low atomic number and as a result a reduced neutron economy, (3) a relatively low boiling temperature of 1165 K, (4) relatively low shielding against gamma-rays and energetic neutrons, (5) producing a radiological hazard when irradiated (Jevremovic et al 1996), (6) solidification at a temperature significantly greater than room temperature (371 K), and (7) a relatively large volume change upon solidification.

Pb-Bi coolant is superior to sodium coolant in some aspects and inferior in other aspects. Pb-Bi coolant has the advantages of (1) chemical inertness with air and water and thus compatible with a simplified containment structure, (2) high boiling temperature (1998 K) and the potential for fuel rod cooling even during temperature excursions, (3) high atomic number and as a result a good neutron economy, (4) relatively high shielding against gamma-rays and energetic neutrons, and (5) a relatively small volume change upon solidification. Pb-Bi coolant has the disadvantages of (1) being potentially corrosive to structural materials, (2) based on technology that is not well established, (3) solidification at a temperature significantly greater than room temperature (398 K), (4) producer of a radiological hazard ($Po^{210}$) when irradiated, and (5) relatively high material costs.

Both the S-PRISM and STAR-LM reactors have reactor cores consisting of an array of fuel rods. The fuel in the S-PRISM reactor may be composed of either oxidic or metallic fuel. A harder neutron spectrum and higher burnups may be achieved with metallic fuel than oxidic fuel. The fuel is clad with a ferritic alloy in order to minimize swelling associated with a high neutron fluence. The fuel in the STAR-LM reactor may be either metallic or nitride in composition. The fuel in the STAR-LM reactor is clad with stainless steel.

Both the S-PRISM and STAR-LM reactors have passive decay heat removal systems. These systems are designed to transport decay heat from the reactor core to the reactor vessel by natural circulation of the coolant in the reactor vessel and then transport the decay heat to the containment vessel and beyond by natural circulation of air and inert gases placed in the gap between the reactor and containment vessels.

In addition to the typical accident initiating events such as pipe breaks, loss of off-site power, loss of generator load, and ejection of control rods, there are several safety concerns unique to liquid metal cooled reactors. These safety concerns include (1) freezing of the coolant in an event causing increased heat removal from the primary coolant, (2) ingress of water into the primary coolant system after rupture of a steam generator tube, and (3) power excursions caused by voids in the primary coolant system and the positive void reactivity coefficient of these reactors. While the SCDAP/RELAP5/3D code has the capability to model most of the phenomena occurring in a liquid-metal cooled reactor after an accident initiating event (MacDonald and Todreas 2000), nevertheless some features of these reactors and some phenomena occurring in these reactors cannot be adequately represented by the code. Improvements to the code required to adequately model liquid-metal cooled reactors include: (1) addition of material properties for the fuel (Zr-Pu, U-PuN and UN) and structural properties of the cladding (stainless steel), (2) modeling of fission gas release in the fuel, (3) modeling of the freezing of liquid-metal, and (4) modeling of inter-component and interphase heat and momentum transfer in a mixture of metallic coolant, liquid water and steam.

## 4.3 Non-Classical Reactors

Although relatively unproven, Molten Salt Reactors (MSRs) are candidates for Generation IV reactors because they offer the possibility for extremely safe production of electricity as well as having proliferation resistant attributes (Robertson 1971, Gat and Dodds 1997, Gromov 1997, Mitachi et al 1999, Vergnes et al 2000). In one design of the MSR, the primary working fluid in the reactor contains the fuel. The Th-U fuel is homogeneously mixed in the fluoride salt working fluid. Fission products can be continuously extracted from the working fluid so as to maintain a very low decay heat level and small radiological source term. In another design of the MSR, coated fuel particles are configured as rods inside graphite blocks (Vergnes 2000). Advantages of the MSR include a high thermal efficiency (44%) and the burning of actinides (Gat and Dodds 1997). The high thermal efficiency of the molten salt reactors is due to the high core exit temperature of the working fluid. In one design of the molten salt reactor, energy is extracted from the working fluid in a tertiary loop containing water above the critical pressure (Robertson 1971), while in another design the energy is extracted in a secondary helium loop (Vergnes 2000). A safety concern for molten salt reactors is freezing of the primary coolant and blockage to coolant flow caused by the freezing.

The gas core reactors have unique safety features. Any loss of system pressure results in a loss of reactivity. Any possible damage due to an accident is limited to reactor pressure vessel and reflector. The low thermal conductance of the gas fuel/working fluid allows for very high average bulk fluid temperatures (greater than 2500 K), while maintaining cool wall temperatures. The use of magnetic turbines requires the modeling of the ionizing of the gaseous fuel by fission and the affect of the magnetic field in the reactor's magnetic turbine on the behavior of the gas fuel/working fluid.

While the SCDAP/RELAP5/3D code has the capability to model some of the phenomena occurring in non-classical reactors after an accident initiating event, nevertheless some features of these reactors and some phenomena occurring in these reactors cannot be adequately represented by the code. The features and phenomena that cannot be adequately represented are (1) in molten salt reactor, a working fluid composed of a salt (2) volumetric heating of the primary working fluid due to the fissile isotopes in the fluid, (3) in molten salt reactors, freezing of the primary working fluid and blockage to flow caused by freezing, (4) in the gas core reactors, ionizing of the gas fuel/working fluid, and (5) affect on behavior of the gas fuel/working fluid of the magnetic field in the magnetic turbine. The required improvements to the SCDAP/RELAP5/3D code, therefore, include the addition of thermal and transport properties for the salts of the molten salt reactors and the gaseous fuel of the gas core reactors. The required improvements also include the modeling of the freezing of the salts for the molten salt reactors. For the gas core reactors, they also include the ionizing of the gaseous fuel and the affect of a magnetic field on the behavior of the gaseous fuel.

## 4.4 Light Water Reactors

While LWR candidates for Generation IV reactors have some common characteristics, such as a primary coolant composed of water and a reactor core with of $UO_2$ fuel in rod-like geometry, nevertheless other characteristics may vary significantly. Some of the differences in proposed Generation IV LWR designs are shown in Table 1. Some of the LWR candidates have the primary coolant loop within the reactor vessel so as to eliminate the possibility of a loss-of-coolant accident and reduce the probability of other accident sequences. The use of natural convection as a driving force for the primary coolant is used in some of the designs to reduce reliance on pumps. A direct thermal cycle in which the coolant from the core outlet flows directly to the electricity producing turbines is used by some of the LWR designs, while other designs use a steam generator to heat the fluid that drives the turbines. Some of the LWR candidates for Generation IV reactors have a containment system that enhances passive cooling. Some of the candidates have a tight lattice reactor core so as to produce a hard neutron spectrum that burns actinides and extends the reactor core lifetime, and thus reduces radioactive waste and decreases operating cost. The cost of initial investment is less for a small LWR than a large LWR; as a result, some of the LWRs are designed as small in size as 35 MWe. Two of the candidates operate with a primary coolant system pressure above water's critical pressure of 22.1 MPa so as to reduce cost by eliminating steam generators and increasing thermal efficiency by operating at a higher coolant outlet temperature. Most of the candidates have active emergency cooling systems in addition to passive cooling systems.

The safety of LWR versions of Generation IV reactors has been enhanced by changes in the configuration of the primary coolant system. Those LWRs with primary coolant loops within the reactor vessel (IRIS and MASLWR) eliminate the potential for loss-of-coolant accidents caused by a break in the primary coolant system piping. These primary coolant systems, therefore, require less containment capacity, allowing for a smaller containment design. The spacing of components in the reactor core has been adjusted in some of the LWRs (IRIS, MASLWR, SBWR) so a significant part or all of the circulation of the primary coolant is driven by natural convection. These design changes prevent or reduce the probability of accidents initiated by events such as a locked pump rotor or loss of pump power. Natural circulation is enhanced by adjustments such as (1) increasing the spacing between fuel rods, (2) decreasing the height of reactor core, (3) decreasing fuel rod power so sufficient cooling and reduced flow losses can be achieved with a reduced core flow rate, (4) increasing the extent of boiling in the reactor core, (5) increasing the flow path between the downcomer and lower plenum, and (6) increasing the height of the reactor vessel to enhance the driving head for natural circulation. The possibilities for uncovery of the reactor core have been reduced in some of the LWRs by increasing the inventory of water above the fuel rods. In the SPWR and CANSP, accidents initiated by a failure in the steam generator have been eliminated by removing this component from the reactor system.

The safety of candidate Generation IV LWRs has also been increased by changes to the configuration of the reactor containment. In general, the containment is designed so decay heat can be removed in a passive manner with simple heat exchangers. The containment may include large pools of water that condense steam to reduce the pressure load on the containment. In general, the containment is inerted with nitrogen to prevent hydrogen deflagration. In the case of the IRIS, the containment is pressurized to reduce the pressure differential that drives coolant out of a break and into the containment (Carelli et al 2001). In the case of one version of the SBWR, the containment has been designed with sufficient capacity to accommodate the hydrogen produced by oxidation of 100% of the reactor core (Brettschuh 2001).

The designs of the SPWR and CANSP are based on the behavior of water above the critical pressure (22.1 MPa). Water at the supercritical pressure of 25 MPa circulates through the reactor core and power generating turbines (Oka et al 1995, Jevremovic et al 1996, Kitoh et al 1998, Mukohara et al 1999, Bushby et al 2000, MacDonald et al 2001a). For water above the critical pressure, the concept of boiling does not exist. This is because water above the critical pressure is a single-phase fluid with no discernible boundary between liquid and gas phases. While a deterioration in heat transfer may occur in a certain range of bulk fluid enthalpy, there are no occurrences of a sharp discontinuity in heat transfer due to dryout. The specific heat exhibits a peak at the pseudo-critical temperature (~658 K). A plot of specific heat as a function of temperature is shown in Figure 5. The temperature of the coolant varies from about 580 K at the core inlet to about 750 K at the core outlet. Since convective heat transfer at supercritical pressure is proportional to specific heat to the 0.4 power, the heat transfer coefficient is large for temperatures around the pseudo-critical temperature (Oka 1995). The heat transfer and enthalpy change with respect to temperature in the range of the pseudo-critical temperature is so large that much heat is removed with low coolant flow. The density change with respect to temperature is also large around the pseudo-critical temperature. A plot of density as a function of temperature for a pressure of 25 MPa is also shown in Figure 5.

Since the SCDAP/RELAP5/3D code was specifically developed to model LWRs, most of the phenomena occurring in Generation IV LWRs can be adequately modeled with the current version of the code. However, because of some of the unique design features described above, some improvements to the code are needed to address the full spectrum of Generation IV LWR designs. These improvements include: (1) the addition of material properties for mixed oxide fuel, metallic fuel, and Ni-based alloy fuel cladding, and models for the corrosion and swelling of Ni-based alloy cladding, (2) model for fuel pellet-cladding mechanical interactions, (3) heat transfer correlations for water above the supercritical pressure, and (4) the capability to interface with a code to perform sub-channel thermal hydraulic analyses.

## 5. SUMMARY OF NEW CAPABILITIES REQUIRED FOR TRANSIENT ANALYSES OF GENERATION IV REACTORS

The transient analyses of Generation IV reactors require advancements in the current capabilities of SCDAP/ RELAP5/3D as described in the previous sections. This section summarizes these required advancements. A planned series of relatively small development efforts applied to the SCDAP/RELAP5/3D code will result in a gradual expansion in the range of designs of Generation IV reactors that can be analyzed. Table 3 summarizes the planned modeling improvements to the SCDAP/RELAP5/3D code, and identifies the Generation IV reactor concepts that these improvements are applicable to.

As indicated in Table 3, the capability for the transient analyses of HTGRs will require improvements primarily in the modeling of the reactor core and the capability for the transient analyses of metal-cooled reactors will be achieved primarily by improvements in the modeling of the coolant thermal-hydraulic behavior. The HTGRs have reactor core compositions and configurations radically different from those of current generation LWRs. As a result, extensions are required to model (1) convective heat transfer from fuel pebbles, (2) oxidation of fuel pebbles, and (3) heat transfer by conduction, radiation, and natural convection through a bed of fuel pebbles. The latter extension in modeling is required for the analysis of transients in which forced flow through the reactor core is lost.

The liquid metal cooled reactors have the possibility of the interaction of the metal coolant with water after an initiating event such as a rupture in a steam generator tube. As a result, extensions are required in the modeling of this thermal-hydraulic behavior. In particular, extensions are required to model the heat and momentum transfer occurring in a mixture of metal coolant and water. Since the coolant in Pb-Bi cooled reactors has a relatively high freezing temperature, extensions are also required to model the freezing of this coolant after initiating events causing an overcooling of the primary coolant.

Finally, the focus of modeling improvements for the advanced LWRs is primarily on extended heat and mass transport properties for water at a pressure above the critical pressure. For the non-classical reactors, the modeling improvements focus on modeling the behavior of the unique primary working fluids.

## 6. CONCLUSIONS

Plans are being developed for extending SCDAP/RELAP5/3D for the transient analyses of each major category of proposed Generation IV reactors. The code already have the fundamental modeling capabilities for the transient analyses of Generation IV reactors. A series of relatively small development efforts to address the unique features of Generation IV reactors achieves the basic capability to perform the transient analyses of a broad range of proposed Generation IV reactors. These development efforts achieve the capability to model (1) transport of heat in reactor cores composed of fuel/graphite pebbles and fuel/graphite blocks, (2) thermal, chemical, and mechanical behavior of fuel rods composed of advanced design fuels and clad with stainless steel or a Ni-based alloy, (3) freezing of working fluid, (4) convective heat transfer in water at a pressure greater than the critical pressure, (5) thermal and transport properties in working fluids composed of salts or $UF_4$, and (6) ionizing of the working fluid and the affect of a magnetic field on the working fluid. The categories of reactors that can be analyzed with the completion of these development efforts include (1) advanced light water reactors including supercritical pressure water reactors, (2) pebble bed and block-type high temperature gas cooled reactors, (3) sodium cooled and Pb-Bi cooled reactors, and (4) non-classical reactors such as the molten salt and gas core reactors.

**Table 3. Extensions in modeling capability required for basic transient analysis of broad range of Generation IV reactors.**

| Extension in modeling capability | Reactor design requiring extension | | | | | | |
|---|---|---|---|---|---|---|---|
| | IRIS | PB-HTGR | BT-HTGR | HMFR | S-PRISM | SPWR, CANSP | MSR, GSR |
| Thermal and fission gas behavior in fuel composed of mixture of $UO_2$ and $PuO_2$ | x | | | | | x | |
| Sub-channel thermal hydraulic analysis | x | | | | | | |
| Transport of heat, flow losses, and oxidation in reactor core composed of graphite pebbles with coated fuel particles | | x | | | | | |
| Transport of heat in reactor core composed of rods with coated fuel particles inserted in graphite blocks | | | x | | | | |
| Multidimensional fluid behavior in reactor core or jet of gas resulting from pipe break | | x | x | | | | |
| Ingress of air/water into reactor vessel after break in pipe or other component | | x | x | | | | |
| Multi-dimensional heat transfer through complex shaped system | | x | | | | | |
| Thermal, chemical, structural, and fission gas behavior in metallic fuel clad with stainless steel | | | | x | x | | |
| Freezing of coolant | | | | x | | | x |
| Heat and momentum transfer in mixture of Pb-Bi and water or Na and water | | | | x | x | | |
| Convective heat transfer, p>22.1 MPa | | | | | | x | x |
| Structural behavior of Ni-based fuel rod cladding | | | | | | x | |
| Corrosion of surfaces contacted by supercritical pressure water | | | | | | x | |
| Fuel-cladding mechanical interaction | | | | | | x | |
| Thermal and transport properties of fuel, volumetric heat generation in fluid field | | | | | | | x |
| Thermal and mechanical properties of materials in structures containing primary working fluid | | | | | | | x |
| Modeling of ionizing of working fluid by fissioning of the fuel and affect on working fluid of magnetic field in magnetic turbines | | | | | | | x |

Plans are also being developed for interfacing other codes and models with SCDAP/RELAP5/3D so as to achieve a complete front to back capability for analysis of Generation IV reactors. These interfaces achieve the capability to integrate calculations of multidimensional fluid behavior in certain regions of the reactor system with the SCDAP/RELAP5/3D calculations of system-wide fluid behavior. These interfaces also achieve the capability to efficiently calculate the radiological consequences of the transient behavior of Generation IV reactors. Until acceptance limits on various aspects of reactor behavior, such as maximum fuel temperature, have been established for each category of Generation IV reactor, the safety of these reactors can be assessed by comparing calculations of radiological consequences in these reactors with the acceptance limits on radiological consequences established for the current generation reactors.

Acknowledgements

Work supported by U. S. Department of Energy under DOE Idaho Field Office Contract DE-AC07-99ID13727.

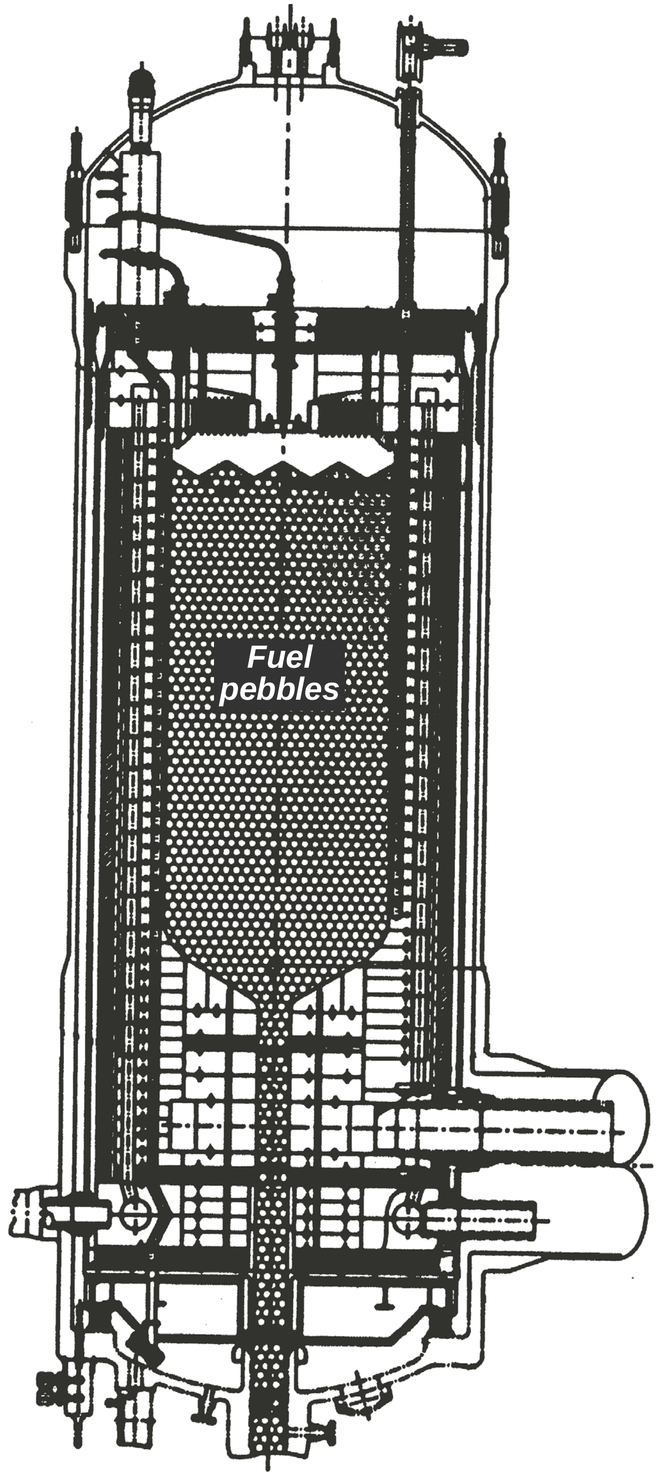


**Figure 1. Reactor Vessel for Pebble Bed High Temperature Gas Reactor.**

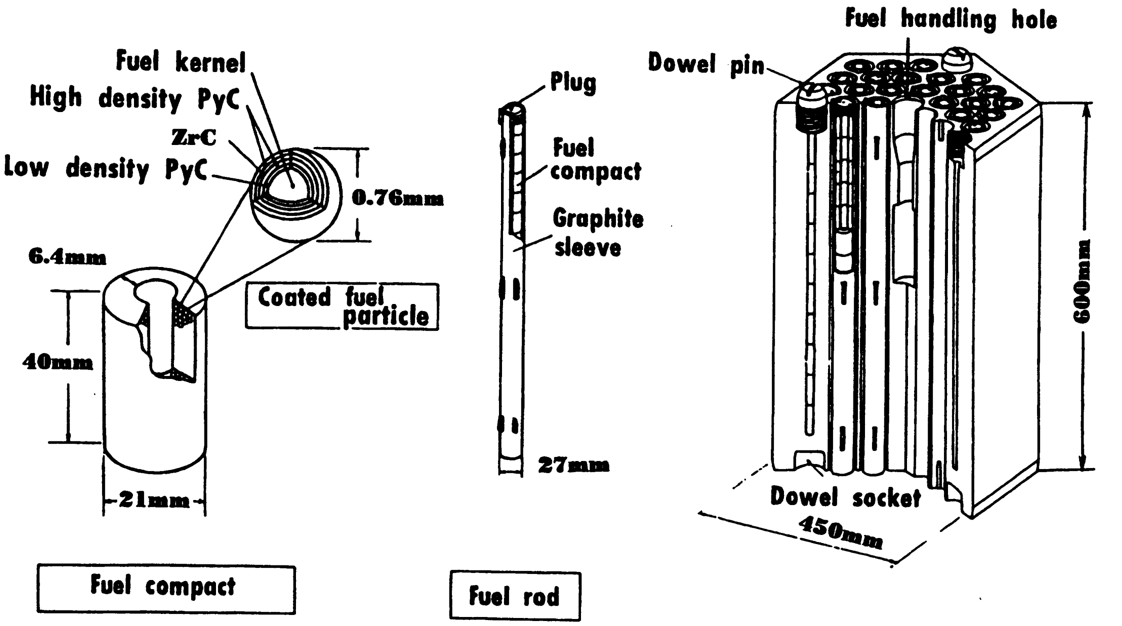


**Figure 2. Configuration of Reactor Core for Block-type High Temperature Gas Reactor.**

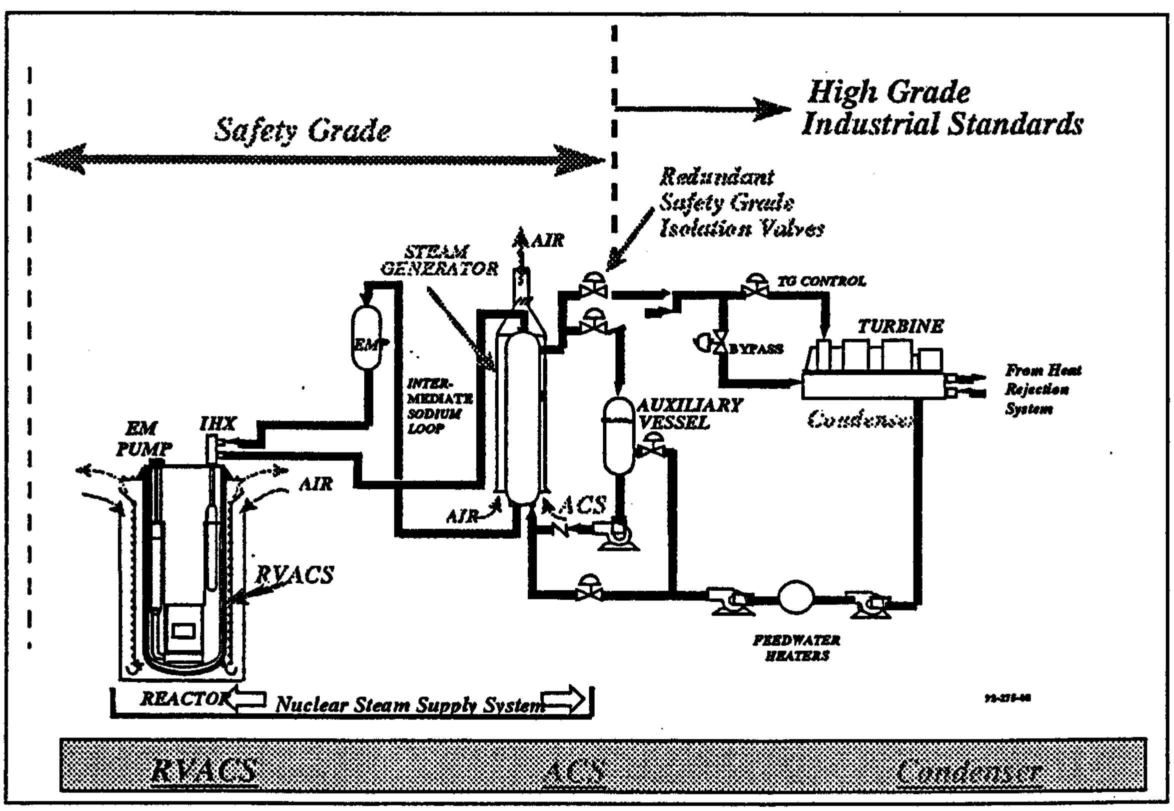


**Figure 3. Schematic of S-PRISM Sodium Cooled Reactor**

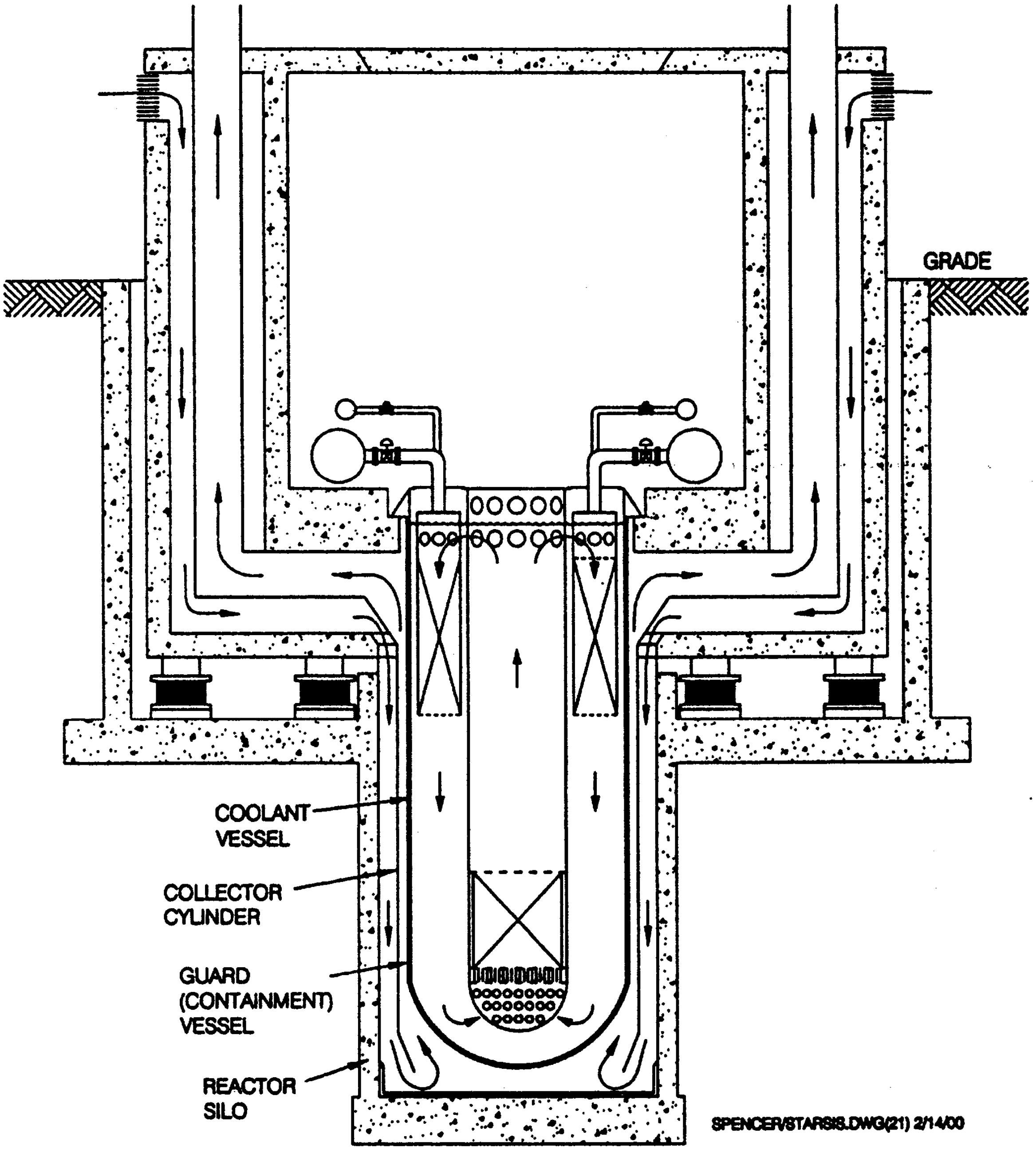


**Figure 4. Cross Section View of the STAR-LM Pb-Bi Cooled Reactor.**

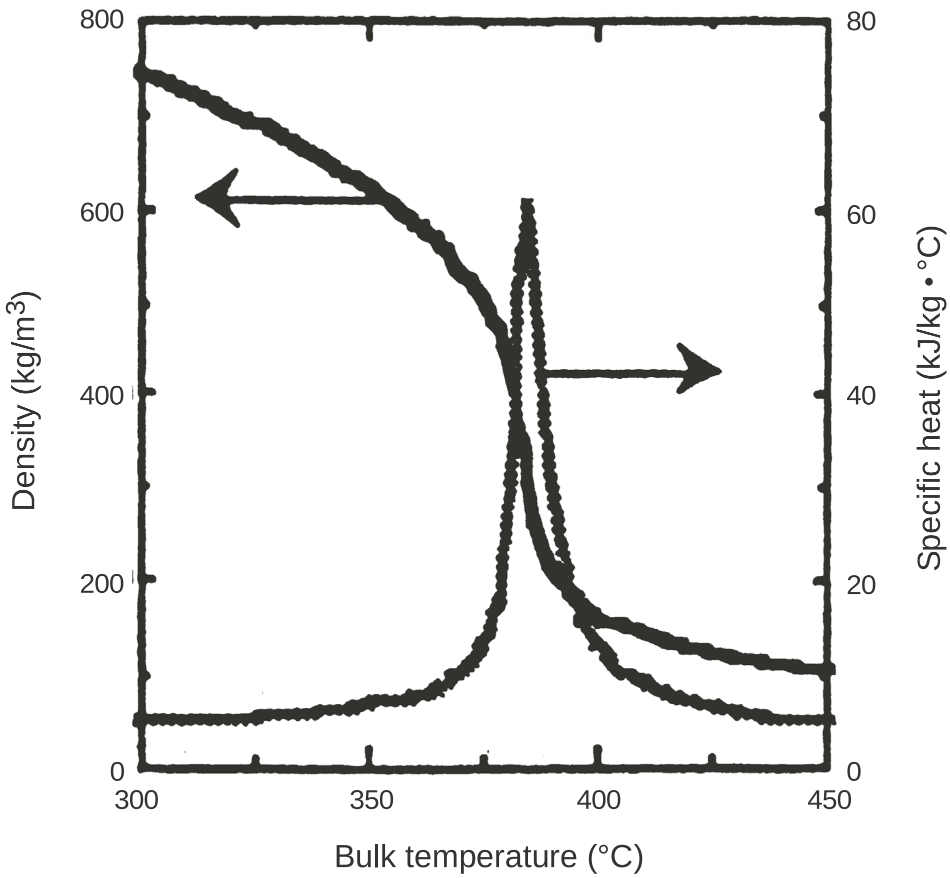


**Figure 5. Density and specific heat as function of temperature for water at pressure of 25 MPa.**

## REFERENCES


S. Anghaie and D. Lewis, “A Summary of Generation IV Non-Classical Power Reactor Concepts,” Transactions of the American Nuclear Society, Nov. 11-16, Reno, Nevada, 2001.

C. E. Boardman et al., “A Description of the S-PRISM Plant,” ICONE-8447, 8th International Conference on Nuclear Engineering, Baltimore MD, April 2-6, 2000.

S. J. Bushby G. R. Gimmick, R. B. Duffy, K. A. Burrell, and S. W. Chan, “Conceptual Designs for Advanced, High-Temperature CANDU Reactors,” Proceedings of ICONE-8, Baltimore MD (April 2000).

W. Brettschuh, “The SWR 1000: A Nuclear Power Plant Concept with Boiling Water Reactor for Maximum Safety and Economy of Operation,” 9th International Conference on Nuclear Engineering, Nice, France, April 2001.

H. L. Brey (Ed.), “Current Status and Future Development of Modular HTGR Technology”, IAEA-TECDOC-TBD, Vienna, 2000.

H. L. Brey, “Historical Background and Future Development of the High Temperature Gas Cooled Reactor,” Seminar on HTGR Applications and Development, Institute of Nuclear Engineering Technology, Singh University, Beijing, China, March 19-21, 2001.

M. D. Carelli, et al., “IRIS, International New Generation Reactor,” ICONE-8447, 8th International Conference on Nuclear Engineering, Baltimore MD, April 2-6, 2000.

M. D. Carelli, et al., “IRIS Reactor Development,” 9th International Conference on Nuclear Engineering, Nice, France, April, 2001.

E. W. Coryell, E. A. Harvego, and L. J. Siefken, *SCDAP/RELAP5/3D Code Manual*, INEEL-EXT-01-00917, July 2001.

DOE, “*Evaluation of the Gas Turbine Modular Helium Reactor”*, DOE-GT-MHR-100002, Gas-Cooled Reactor Associates, San Diego, CA, February 1994.

C. J. Freitas, “Perspective: Selected Benchmarks from Commercial CFD Codes,” Journal of Fluids Engineering, Volume 117, June 1995, pages 208-218.

U. Gat and H.L. Dodds, “Molten Salt Reactors – Safety Options Galore, CONF-970649-6.

J. H. Gittus, “The ESKOM Pebble Bed Modular Reactor,” Nuclear Energy, 38, No. 4, Pages 215-221, August 1999.

IAEA, “International Cooperation in Developing the GT-MHR,” IAEA-TECDOC-988, Pages 59-71, Vienna, December, 1997.

M. Ishii, Modular and Full Size Simplified Boiling Water Reactor Design with Fully Passive Safety Systems, U. S. Department of Energy, Nuclear Energy Research Initiative, Abstract, Proposal No.: 99-0097, 1999.

T. Jevremovic, Y. Oka, and S. Koshizuka, “$UO_2$ Core Design of a Direct-Cycle Fast Converter Reactor Cooled by Supercritical Water,” Nuclear Technology, Vol. 114, pp. 273-284, June 1996.

C. Kadak, “The MIT Modular Pebble Bed Reactor Project,” Seminar on HTGR Applications and Development, Institute of Nuclear Engineering Technology, Tsinghua University, Beijing, China, March 19-21, 2001.

K. Kitoh, S. Koshizuka, and Y. Oka, “Pressure- and Flow-Induced Accident and Transient Analyses of a Direct-Cycle, Supercritical-Pressure, Light-Water-Cooled Fast Reactor,” Nuclear Technology, Vol. 123, Pages 233-244, September 1998.

K. Kunitomi, et al., “Conceptual Design of a 50-MW Severe-Accident-Free HTR and the Related Test Program of the HTTR,” Nuclear Technology, Vol. 123, September 1998.

P. E. MacDonald and N. Todreas, *Design of an Actinide Burning, Lead or Lead-Bismuth Cooled Reactor that Produces Low Cost Electricity, Annual Project Status Report*, MIT-ANP-PR—71, INEEL/EXT-2000-00994, July 2000.

P. MacDonald, J. Buongiorno, C. Davis, K. Weaver, G. Was, L. Fournier, M. Carelli, D. Paramonov, and L. Conway, *Feasibility Study of Light Water Cooled Fast Reactors for Actinide Burning and Electric Power Production*, Nuclear Energy Research Initiative Proposal Program Announcement LAB NE-2001-1, 2001

P. E. MacDonald and K. R. Hedges, “Generation IV Water-Cooled Reactor Concepts,“ Transactions of the American Nuclear Society, Nov. 11-16, Reno, Nevada, 2001.

R. K. McCardell et al, *NPR-MHTGR Fuel Development Program Plan*, EGG-NPR-8971, September 1990.

C. McNeill, “A PBMR in Exelon’s future?” Nuclear News, (February 2001).

G. Mesina and J. Galbraith, “New Developments and Value of the RELAP5-3D Graphical User Interface (RGUI 1.2), Idaho National Engineering and Environmental Laboratory, Proceedings of the RELAP5-3D International User Seminar, Jackson Hole, WY, September 12-14, 2000.

K. Mitachi et al., “A Preliminary Design Study of a Small Molten Salt Reactor for Effective Use of Thorium Resource,” 7th International Conference on Nuclear Engineering, Tokyo, Japan, April 19-23, 1999.

S. M. Modro, et al., *Multi-Application Small Light Water Reactor*, DOE Nuclear Energy Research Initiative, Baseline Concept Report, Idaho National and Engineering Laboratory, October 2000.

T. Mukohara, S. Koshizuka, and Y. Oka, “Core Design of a high-temperature fast reactor cooled by supercritical light water,” Annals of Nuclear Energy 26, Pages 1423-1436, 1999.

Y. Oka, S. Koshizuka, T. Jevremovic, and Y. Okano, “Systems Design of Direct-cycle Supercritical-Water-Cooled Fast Reactors,” Nuclear Technology, Vol. 109, Pages 1-10, Jan. 1995.

S. Rao and A. Gonzalez, “ESBWR Program - Development of Passive Plant Technologies and Designs,” ICONE-8205, 8th International Conference on Nuclear Engineering, Baltimore MD, April 2-6, 2000.

The RELAP5 Development Team, RELAP5/MOD3 Code Manual, Vol. I: Code Structure, System Models, and Solution Methods, NUREG/CR-5535, INEL-95/0174, August 1995.

The RELAP5-3D Code Development Team, RELAP5-3D Code Manual, Volume I: Code Structure, System Models and Solution Methods, INEEL-EXT=98-00834, Revision 1.3a, February 2001.

R. C. Robertson, *Conceptual Design Study of a Single Fluid Molten-Salt Breeder Reactor*, ORNL-4541, June 1971.

S. L. Rosen, Y. Sagayama, and M. J. Lineberry “Report on Generation IV Technical Working Group 3: Liquid Metal Reactors,” Transactions of the American Nuclear Society, Nov. 11-16, Reno, Nevada, 2001.

R. R. Schultz, *Generation IV Gas Cooled Reactor Systems Analysis* Tools, Idaho National Engineering and Environmental Laboratory LDRDGE103.

H. Sekimoto and Z. Su’ud, “Design Study of Lead and Lead-Bismuth Cooled Small Long-Life Nuclear Power Reactors Using Metallic and Nitride Fuel,” Nuclear Technology, Vol. 109, March 1995.

L. J. Siefken, E. W. Coryell, E. A. Harvego, and J. K. Hohorst, *SCDAP/RELAP5/MOD3.3 Code Manual: Modeling of Reactor Core and Vessel During Severe Accidents*, NUREG/CR-6150, Vol. 2, Rev. 2, INEL/96/0422, January 2001.

F. Southworth, F. Carre, and P. Hildebrant, “Generation IV Gas Cooled Reactor Concepts,” Transactions of the American Nuclear Society, Nov. 11-16, Reno, Nevada, 2001.

B. W. Spencer et al., “An Advanced Modular HLMC Reactor Concept Featuring Economy, Safety, and Proliferation Resistance,” ICONE-8, 8th International Conference on Nuclear Engineering, Baltimore MD, April 2-6, 2000.

R. M. Summers et al., *MELCOR 1.8.0: A Computer Code for Severe Nuclear Reactor Accident Source Term and Risk Assessment Analysis*, NUREG/CR-5531, SAND90-0364, Sandia National Laboratories, January 1991.

H. A. Upton, F. E. Cooke, and J. K. Sawabe, “Simplified Boiling Water Passive Safety Features,” Transactions of the American Nuclear Society, Vol. 68, Part A, June 20-24, San Diego, California 1993.

Nuclear Regulatory Commission, USNRC Standard Review Plan, Office of Nuclear Reactor Regulation, NUREG-0800, July 1981.

Su’ud, Z. and Setimoto, H. “Accident Analysis of Lead or Lead-bismuth Cooled Small Safe Long-term Fast Reactor Using Metallic or Nitride Fuel”, Nuclear Engineering and Design, Vol. 162: 2-3, April 1996.

I. Van Heek, “The Pebble Bed HT, the Nuclear Option in the Netherlands,” Seminar on HTGR Applications and Development, Institute of Nuclear Engineering Technology, Tsinghua University, Beijing, China, March 19-21, 2001.

J. Vergnes et al., “The AMSTER concept (Actinide Molten Salt TransmutER)”, Physor 2000 conference, Pittsburg, Pennsylvania, May 2000.)

K. E. Washington et al., *Reference Manual for the CONTAIN 1.1 Code for Containment Severe Accident Analysis*, NUREG/CR-5715, SAND91-0835, July 1991.

K. D. Weaver, J. S. Herring, and P. E. MacDonald, “Performance Comparison of Metallic, Actinide Burning Fuel in Lead-Bismuth and Sodium Cooled Fast Reactors,” 9th International Conference on Nuclear Engineering, Nice, France, April 2001.

K. Yamashita et al., *Conceptual Design Study of Pebble Bed Type High Temperature Gas Cooled Reactor with Annular Core Geometry*, JAERI-M-90-153, Japan Atomic Energy Research Institute, September 1990.